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NRL/NUREG Memorandum Report 3600

Structural Integrity of Water Reactor Pressure Boundary Components

PROGRESS REPORT ENDING 31 MAY 1977

F. J. Loss, Editor

Thermostructural Materials Branch
Engineering Materials Division

September 1977

Prepared for the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under Contract AT(49-24)-0207

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20. ABSTRACT (Continue on reverse side if necessary and identify by block number)

This report describes research progress in a continuing program to characterize materials properties performance with respect to structural integrity of light water reactor pressure boundary components. Progress for this reporting period is summarized in the following areas: (a) evaluation of critical factors in fatigue crack growth rate studies in a pressurized water reactor environment; (b) fractographic observations of fatigue crack propagation; (c) the IARAR program and its objectives; (d) the effect of submerged arc welding on the notch toughness of low upper shelf A302-B plate

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20. Abstract (Continued)

exhibiting a low upper shelf and its weld HAZ, and (f) surveys of foreign research related to the safety of light water reactors in Japan, United Kingdom, France, and Germany.

CONTENTS

STRUCTURAL INTEGRITY OF WATER REACTOR
PRESSURE BOUNDARY COMPONENTS

PROGRESS REPORT ENDING 31 MAY 1977

SUMMARY

I. FATIGUE CRACK PROPAGATION IN LWR MATERIALS

A. Evaluation of Critical Factors in Crack Growth Rate Studies

Fatigue crack propagation data are presented for A508-2 forging material. Tests were conducted in accordance with the preliminary NRC matrix designed to characterize hydro and leak transients as well as heatup and cooldown transients of a commercial nuclear pressure vessel. To investigate the preceding transients, tests were conducted in 93°C (200°F) reactor grade water at atmospheric pressure and in 288°C (550°F) water at 14 MPa (2000 psig). Results presented here suggest that the test temperatures of 93°C and 288°C have little effect on the data for a wave form consisting of relatively short rise and hold times. Also data compare favorably when the loading time is increased by a factor of 60, from 1 sec to 60 sec. Finally, the combination of a 1 min loading time and a 3 min hold produced a higher crack propagation rate, contrary to expectations.

B. Fractographic Observations on A508 Class 2 Forging

Scanning electron microscopy (SEM) and optical means were used to determine the fatigue crack growth rates of A508 Class 2 forging steel tested in a water environment. The growth rates determined by these two methods were compared to the macroscopic rates obtained from specimen compliance changes in order to establish the degree of correlation between the fractographic measurements and the corresponding mechanical measurements. Good agreement was found between the growth rate from the beach marks (optical) and the corresponding macroscopic rate. The agreement between the striation spacings determined from SEM measurements and the macroscopic rate generally was somewhat less precise than the preceding comparison. However, the

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agreement between SEM and compliance measurements improved with increasing ΔK value. Through the SEM examination it has heen observed that the striation markings on the fracture surfaces were ill-defined and sometimes discontinuous in the region of low crack growth rate (region of low ΔK). The resolution of the markings improved in the region of faster crack growth rate (region of high ΔK). These observations suggest that the method of determining crack growth rate from striation spacings will be restricted to regions where the crack growth rate is greater than approximately 2×10^{-4} mm/cycle (8×10^{-6} in./cycle).

II. RADIATION SENSITIVITY AND POSTIRRADIATION PROPERTIES RECOVERY

A. IARAR Program and Its Objectives

The cyclic irradiation and anneal (IARAR) program devised by NRL and NRC to explore questions of material radiation embrittlement and annealing recovery behavior is described in detail. The experiment matrix includes two A533-B submerged arc welds and one A302-B modified plate and fourteen reactor experiments. The experimental plan is to evaluate both Charpy-V notch ductility and fracture toughness trends with emphasis on upper shelf behavior. The effectiveness of 343°C (650°F) versus 399°C (750°F) heat treatment will be investigated. The present status of the investigations is reported.

B. Effect of Submerged Arc Welding on the Notch Toughness of Low Upper Shelf A302-B Plate

The effect of submerged arc welding on the Charpy-V notch ductility and fracture toughness of an A302-B steel plate having a low Charpy-V upper shelf energy level was explored. The study employed simulated weld HAZ specimens prepared in a Gleeble apparatus. The thermal cycle depicted the use of a welding heat input of 39.5 kilojoules/cm (100 kilojoules/in.).

A significant reduction in Charpy-V upper shelf energy level and transition temperature was observed for the simulated HAZ specimens. Specifically, a 25 percent reduction in upper shelf level and a 56°C reduction in Charpy-V 41J (30 ft-lb) transition temperature was found relative to the unwelded plate condition. In addition, dynamic fracture toughness on the upper shelf was reduced to 129 MPa$\sqrt{m}$ (117 ksi$\sqrt{in.}$).

C. Assessment of Radiation Resistance of A302-B Plate Exhibiting a Low Upper Shelf and Its Weld HAZ

The investigation compares 288°C (550°F) radiation effects to longitudinal versus transverse test orientations of an A302-B plate having highly directional notch ductility properties. The investigation additionally compares the radiation resistance of a simulated weld heat affected zone in the plate TL orientation. Radiation-induced changes in upper shelf energy were found to be small or negligible with 2 x 10^{19} n/cm^2 >1 MeV, consistent with the low copper and phosphorus content of the plate. However, the HAZ appeared much more sensitive to irradiation than the plate in terms of transition temperature elevation. Limited postirradiation fracture toughness data are also presented for the materials.

III. SURVEY OF FOREIGN RESEARCH RELATED TO THE SAFETY OF LIGHT WATER REACTORS

A. Visits to Japanese Research Organizations

In order to maintain abreast of foreign research related to light water reactor safety, visits were made to several Japanese organizations in connection with the 3rd International Conference on Pressure Vessel Technology held in Tokyo. The organizations visited included: Japan Atomic Energy Research Institute, Nippon Steel Research Laboratory, and Japan Steel Works. A meeting of the IAEA Working Group on Reliability of Reactor Pressure Components was also attended. Discussions included fatigue crack propagation, steelmaking practice as well as the forging and welding of reactor pressure vessels.

B. **European Research on Fatigue Crack Propagation in Water**

Discussions were held at several laboratories on the status of research related to cyclic fatigue crack propagation in a reactor water environment. The objectives and approach of the European programs generally parallel those in the USA.

RESEARCH PROGRESS

I. FATIGUE CRACK PROPAGATION IN LWR MATERIALS

A. Evaluation of Critical Factors in Crack Growth Rate Studies

H. E. Watson, B. H. Menke, and F. J. Loss

BACKGROUND

Experimental results discussed here relate to an evaluation of the effects of water, temperature, and cycling rate on the fatigue crack propagation (FCP) in A508-2 forging material. Tests are being conducted in accordance with a preliminary matrix developed by the Nuclear Regulatory Commission (NRC) to simulate: (a) the hydro and leak transient, (b) the heatup and cooldown transient, and (c) the steady state operation of a commercial nuclear pressure vessel. Emphasis is being placed on one material for the subject studies to minimize the effect of material composition on FCP in the preliminary program to define the important variables. The test conditions corresponding to conditions (a) and (b) are, respectively, 93°C (200°F) water at atmospheric pressure and 288°C (550°F) water at a pressure of 14 MPa (2000 psi). Test specimens used in this series are 25 mm (1 in.) thick compact tension (CT). The loading wave forms for these tests included: a modified trapezoid with a load/unload time of 1 sec and a hold time of 1 min; a triangle with a load/unload time of 1 min; and a modified trapezoid with a loading time of 1 min, hold time of 3 min, and unload time of 1 sec. The R-ratio used for all tests was 0.125. Crack length measurements are determined by the compliance method wherein changes in the crack length are computed from changes in the mouth opening (CMO). In all cases, the crack growth rate, da/dN, values were determined by computer analysis using the incremental polynomial technique recommended by the ASTM Task Group on Fatigue Crack Growth Rate Testing (1).

EXPERIMENTAL PROCEDURE

The FCP data were generated using both autoclave and water pot fatigue test equipment to simulate the heatup and hydro transients, respectively. The autoclave is pressurized to 14 MPa (2000 psi) and heated to 288°C (550°F). The autoclave water is circulated through a filter and returned to a reservoir which is pressurized with hydrogen to 0.2 MPa (30 psi). This maintains the dissolved hydrogen at 30 to

50 cc/Kg in accordance with water specifications in a pressurized water reactor (PWR) system. Crack length measurements are referenced to the CMO. Measurement of the CMO in both the autoclave and water pot are determined using a linear variable differential transformer (LVDT) designed to withstand the environment. LVDT measurements are then converted to crack lengths using compliance data obtained from A533-B steel specimens having machined notches of different depths.

The water pot and autoclave environments are identical except for temperature and cover gas. The water pot is maintained at 93°C (200°F) and has an overpressure of nitrogen. The autoclave temperature is 288°C (550°F) and has an overpressure of hydrogen. The water in both test chambers simulates, as nearly as possible, the actual PWR water chemistry. The chemical composition was listed in reference 2. The water in both test chambers is also circulated to maintain a uniform chemical composition.

RESULTS

Results presented in Fig. 1 show the similarity of data obtained in the water pot at 93°C (200°F) and the autoclave at 288°C (550°F). All other test conditions are identical except for the cover gas. The loading wave form for these tests is a trapezoid with a 1 sec load/unload time and a 1 min hold at maximum load. The Section XI air and water upper bound curves are shown as a reference. These data clearly show that there is little effect of the temperature difference for a wave form consisting of relatively short rise and hold times.

To determine the effect of rise time an autoclave test was conducted using a triangular wave form with a one min load/unload time (Fig. 2). A comparison of the data in Fig. 1 and 2, by reference to the Section XI upper bound curves shown in both figures, suggests that there is no apparent effect shown by increasing the ramp time by a factor of 60. Figure 3 presents water pot data generated using a 1 min ramp, 3 min hold, and a 1 sec unload time. The addition of the 3 min hold time on the cycle of Fig. 2 resulted in the highest crack growth rates obtained at NRL in PWR material. This high growth rate could also be the result of data scatter in that other studies (3) have shown an absence of any crack extension during long periods (several months) of static hold time at load. Subsequent data from the NRC preliminary test matrix should resolve this uncertainty; Tests are scheduled to characterize the effect of rise and hold times of up to 30 and 60 min respectively.

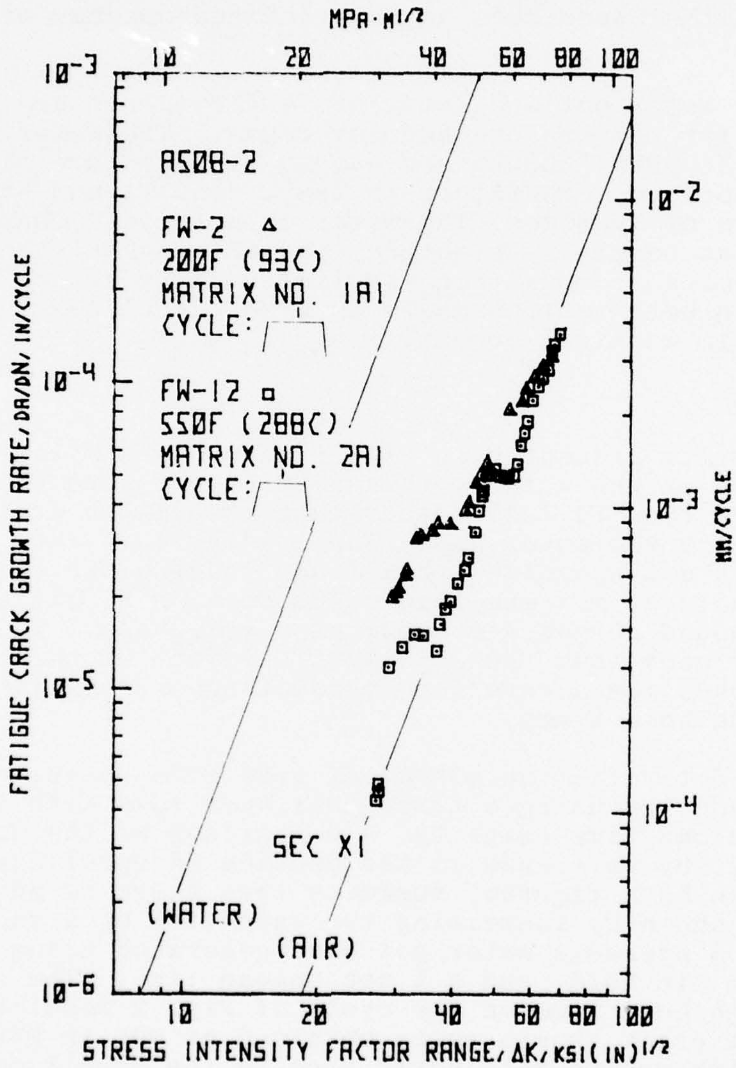

Fig. 1 — Comparison of autoclave (288°C) and water pot (93°C) fatigue crack propagation data. The fatigue cycle is indicated and the points are indexed to the NRC test matrix. ASME Section XI upper bound curves for water and air environments are presented as a reference.

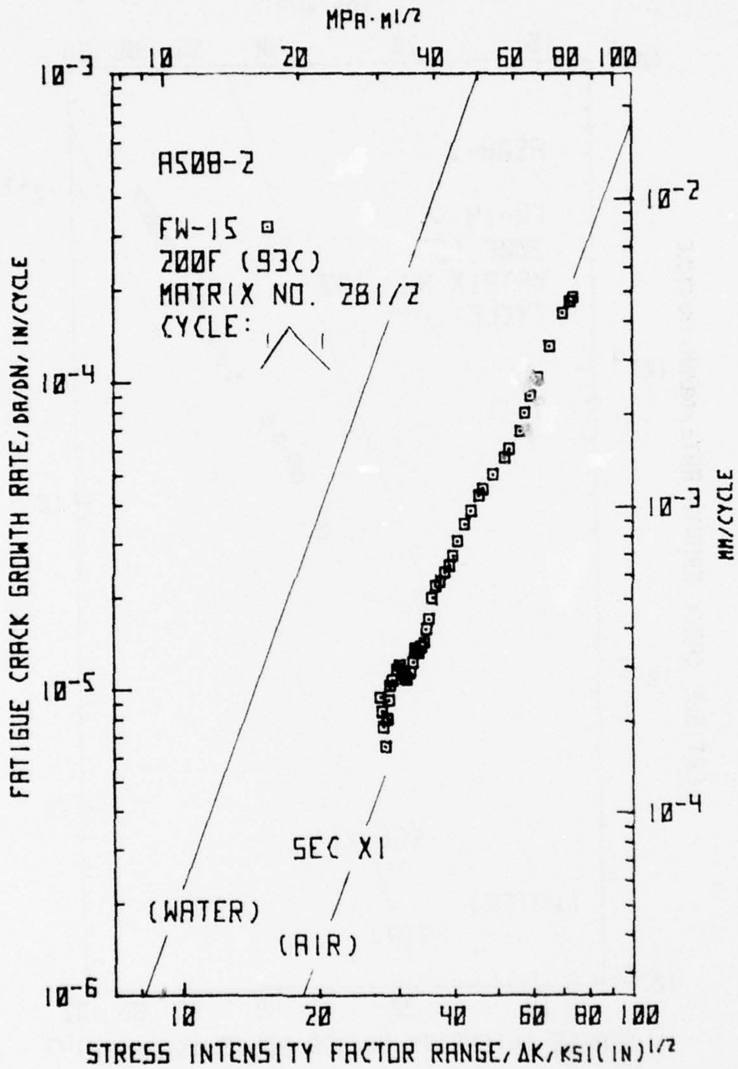

Fig. 2 — Water pot data with a 1 min load/unload (triangular) wave form is referenced to ASME Section XI upper bound data

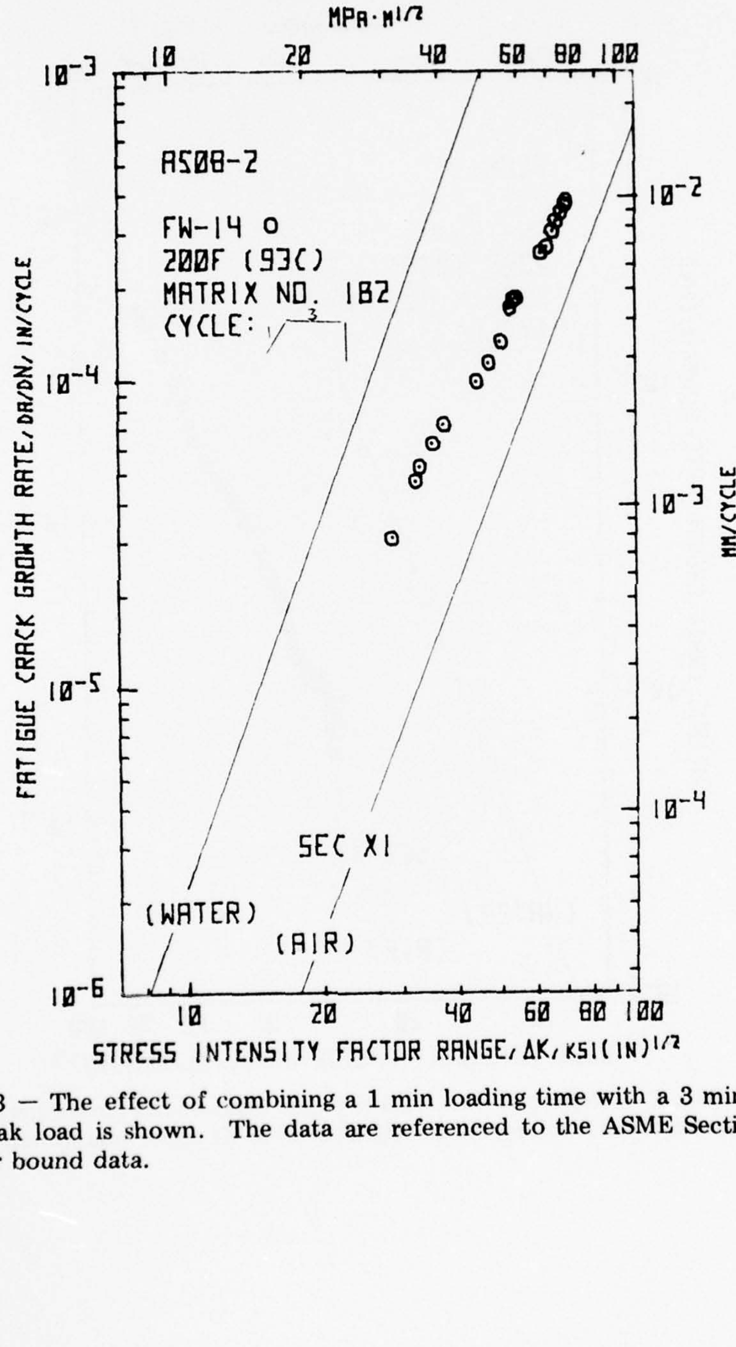

Fig. 3 — The effect of combining a 1 min loading time with a 3 min hold at peak load is shown. The data are referenced to the ASME Section XI upper bound data.

An autoclave test is underway with a 5 min ramp, 1 min hold, and 3 sec unload time. This test has 10,800 fatigue cycles and the data generated to date are illustrated in Fig. 4. It is estimated that approximately 14 weeks will be consumed in completing this test. In Fig. 4, the data shown between the Section XI water and air lines may not be reliable in that this denotes the beginning of crack extension; additional crack extension produced a lower FCP rate.

For comparison purposes, all data discussed above are summarized in Fig. 5. This comparison suggests that temperature in the range from 93°C (200°F) to 288°C (550°F) has no effect on FCP. Also, 1 min ramp data compare very well with data from 1 min hold time tests. Unexpectedly, the combination of a 1 min ramp and a 3 min hold time produce the highest FCP rates obtained in PWR material at NRL. The reasons for this behavior are being investigated.

B. Fractographic Observations on A508 Class 2 Forging

V. Provenzano and F. A. Smidt, Jr.

BACKGROUND

As part of the NRL program to assess the fatigue crack growth rates in pressure vessel steels, compact tension tests are being conducted with A508 Class 2 forging steel at temperatures up to 288°C (550°F) in a reactor water environment. The planned fatigue testing sequence includes tests with long hold times (up to 1 hour). In these long hold time tests, the crack growth per cycle is expected to be quite small and the available instrumentation may not be sensitive enough to detect small increments of crack extension. The present work had three basic objectives. In order of importance they were: (a) to explore the possibility of utilizing scanning electron microscopy (SEM) to accurately determine the crack growth rates from fatigue striation spacings on the specimen fracture surfaces that were formed during periods of small crack extension in order to project fatigue crack growth rate from a limited number of cycles, (b) to study the microstructure with the SEM to gain additional insight about the fatigue crack growth behavior of this material, and finally, (c) to verify the correlation between the crack growth rate determined microscopically from specimen compliance changes and that measured from the beach marks on the fracture surface.

EXPERIMENTAL PROCEDURE

Electron fractographic studies have shown that under

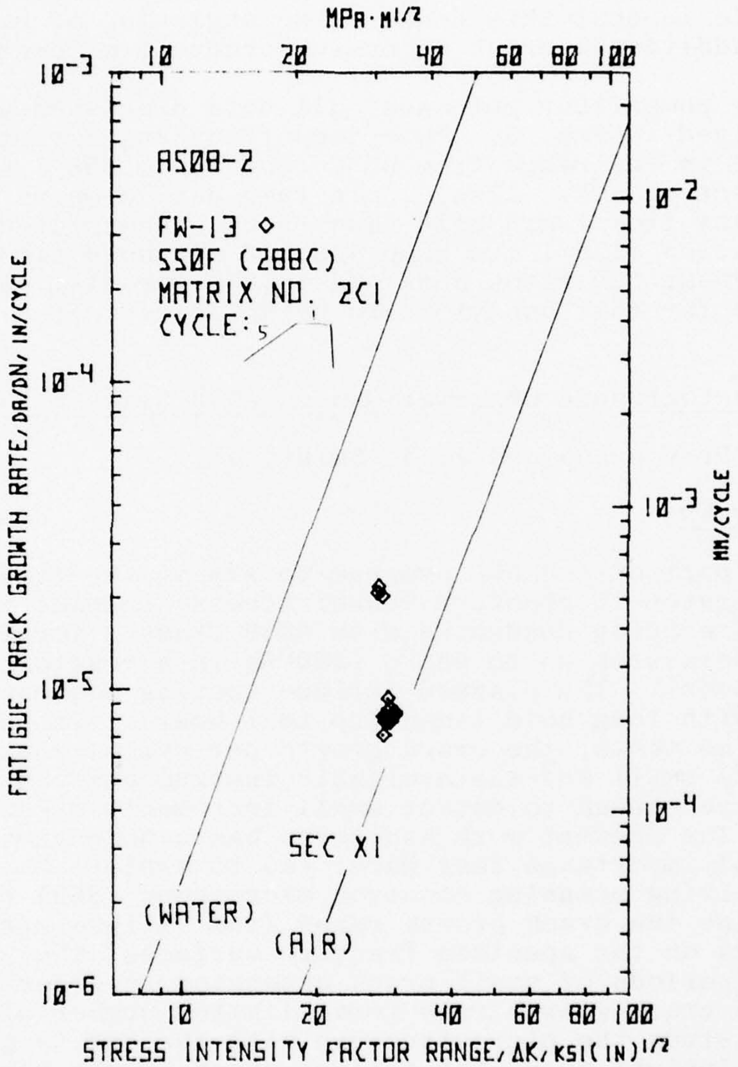

Fig. 4 — Fatigue crack propagation data produced with a loading wave form consisting of a 5 min rise time and a 1 min hold time

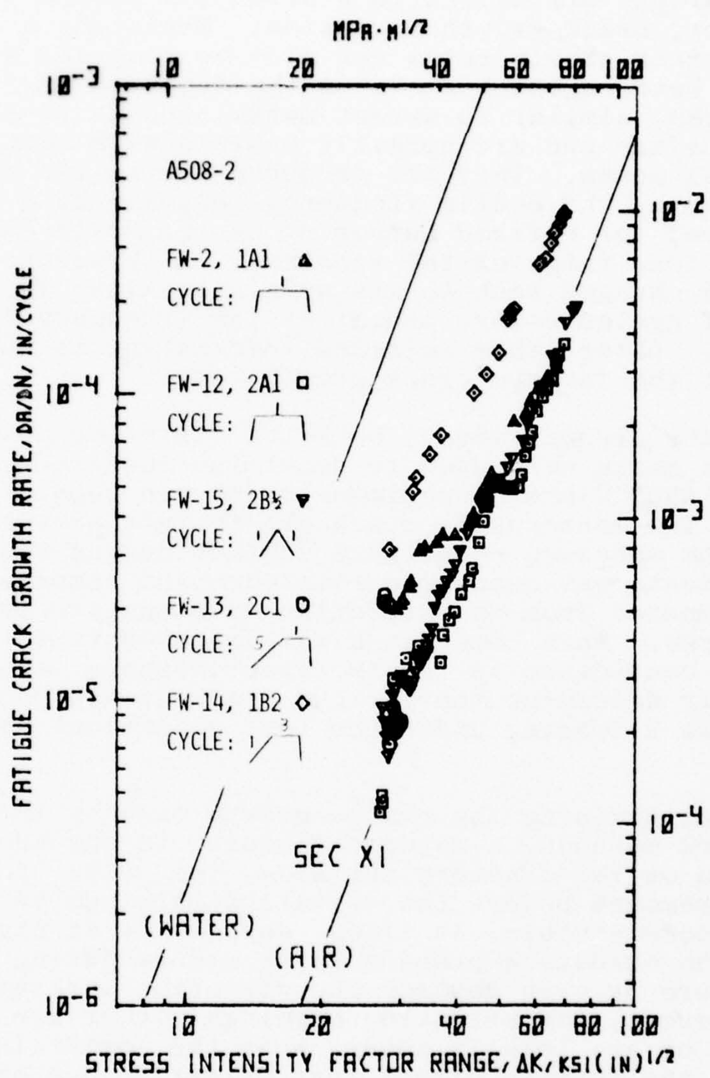

Fig. 5 — Comparison of FCP data illustrated in Figs. 1-4

cyclic loading each fatigue striation observed on the fracture surface corresponds to one load cycle. The spacing of these striations is a measure of the microscopic or local rate of advance of the crack front. Fatigue crack growth rates can be obtained by measuring the average spacing of fatigue striations in a direction parallel to the macroscopic crack growth direction. Employing a different method, crack growth rates can also be computed from the distance between beach marks on the fracture surface. Beach marks, similar to arrest marks, appear as narrow bands on the surface and are normally visible with unaided eye or by optical means. They are produced during the fatigue test by increasing the cyclic frequency (usually by a factor of 10 or more) for a fixed number of cycles while maintaining the same load range on the specimen. Afterwards, the frequency is changed back to its original value. The total number of cycles before and after the frequency change is recorded. Later, this recorded information is used to reconstruct the fatigue crack growth rate.

In the present study, both the striation spacings and the beach marks were used to determine the crack growth rates of A508 Class 2 specimens which had been previously tested. The macroscopic crack growth rate previously computed from specimen compliance changes during the actual fatigue tests was used as a reference and compared with the rates computed from the striation spacings and from the beach marks. This comparison was intended to establish a level of confidence in the two fractographic methods to accurately determine the fatigue crack growth behavior in A508 Class 2 forging under the test conditions described above.

In determining the crack growth rate by the SEM and beach mark methods, a major difficulty is the problem of oxidation on the fracture surfaces. The oxide on the surface must be removed before the striation markings or the beach marks become visible. At times, especially at high temperatures, the oxidation process which occurs during testing can obscure or even destroy the striation markings (4). In this event, the striation markings either are not clearly visible or are totally absent when the oxide is removed. Clearly, the adverse effect that oxidation can have on the formation of well-defined striation markings is a basic limitation on the method for computing crack growth rates from the striation spacings.

RESULTS

The fracture surfaces of A508 Class 2 specimens were examined for fatigue striation markings on a Coates and

and Welter 106A field emission scanning electron microscope (SEM) using the secondary emission mode. It was necessary to remove a heavy oxide layer from the surfaces by a special technique which consisted of two steps. First, the specimen surfaces were ultrasonically cleaned in acetone and in alcohol to remove oily deposits and loose scales. This was followed by immersion in an inhibited HCl solution (5), composed of 100 ml 6N HCl with 0.2 gr hexamethylene-tetramine, to remove the oxide layer. This oxide removal technique does not attack the fracture surface as documented in two recent studies (4,6). All the SEM measurements were made in the mid-thickness section of the specimen along the direction of the macroscopic crack growth rate and were indexed to the corresponding crack length by measuring the distance from the notch with the stage micrometer of the SEM.

Typical SEM micrographs of A508 Class 2 steel alloy that had been fatigue tested at 93°C (200°F) in water are shown in Fig. 6; striation markings can be seen in the micrographs. The two micrographs represent the region of slower crack growth rate (ΔK of 33 MPa$\sqrt{m}$) and the region of faster growth rate (ΔK of 66 MPa$\sqrt{m}$), respectively (Fig. 7). From the micrographs the following general observations about the striation markings can be made.

(1) The fatigue striation markings are more easily resolved in the higher than in the lower ΔK region.

(2) The striation markings are sometimes ill-defined and discontinuous.

(3) The striation spacings exhibit local variation due to changes in orientation of the crack front.

The microscopic crack growth rate was obtained by measuring the average spacing of fatigue striations in a direction parallel to the macroscopic crack growth rate. This growth rate was compared with the macroscopic crack growth rate that had been previously established from specimen compliance changes. Note that an average crack length is used in computing the crack growth rate by the compliance method. However, the crack length index for the striation spacing measurement was determined at the specimen mid thickness. This fact introduces a negligible error (2 to 6%) in the ΔK values associated with the corresponding growth rates when the crack front curvature is slight as in the present investigation. As shown in Fig. 7, good agreement was found between the growth rate determined from the SEM measurements and the macroscopic crack growth rate. A similar correspondence was found between the two growth rates for a specimen tested at the same cyclic rate but in a 288°C (550°F) water environment (Fig. 8).

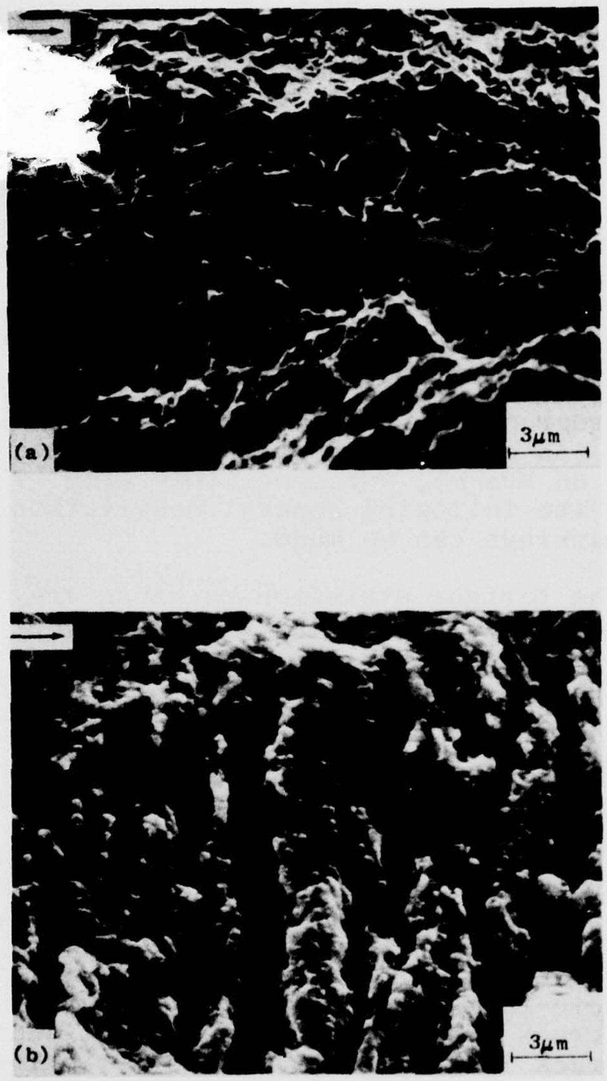

Fig. 6 — Fracture surfaces (SEM) of A508 Class 2 forging steel specimen fatigue tested at 93°C (200°F) in water. The SEM micrographs were taken at crack lengths that correspond to the following ΔK values: (a) $\Delta K = 33$ MPa$\sqrt{m}$ (30 ksi$\sqrt{in.}$), and (b) $\Delta K = 66$ MPa$\sqrt{m}$ (60 ksi$\sqrt{in.}$) arrow indicates macroscopic crack propagation direction. Striation markings can be seen in the micrographs.

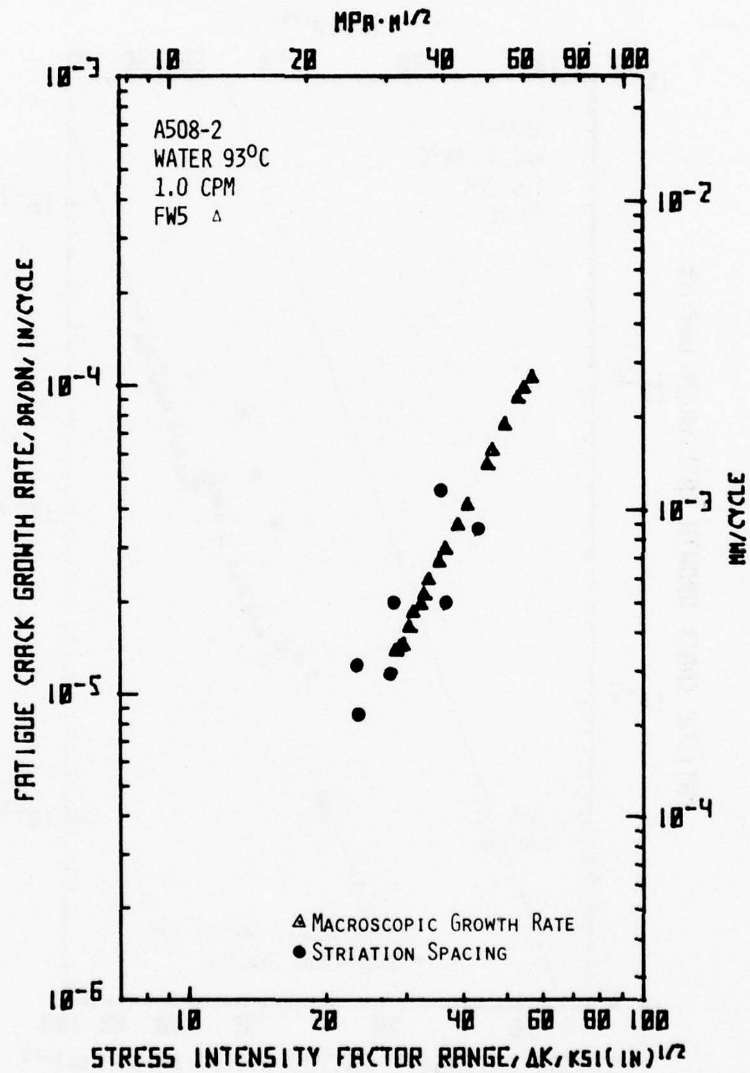

Fig. 7 — Comparison of macroscopic growth rate and striation spacing of A508 Class 2 forging steel fatigue tested at 93°C (200°F) in water

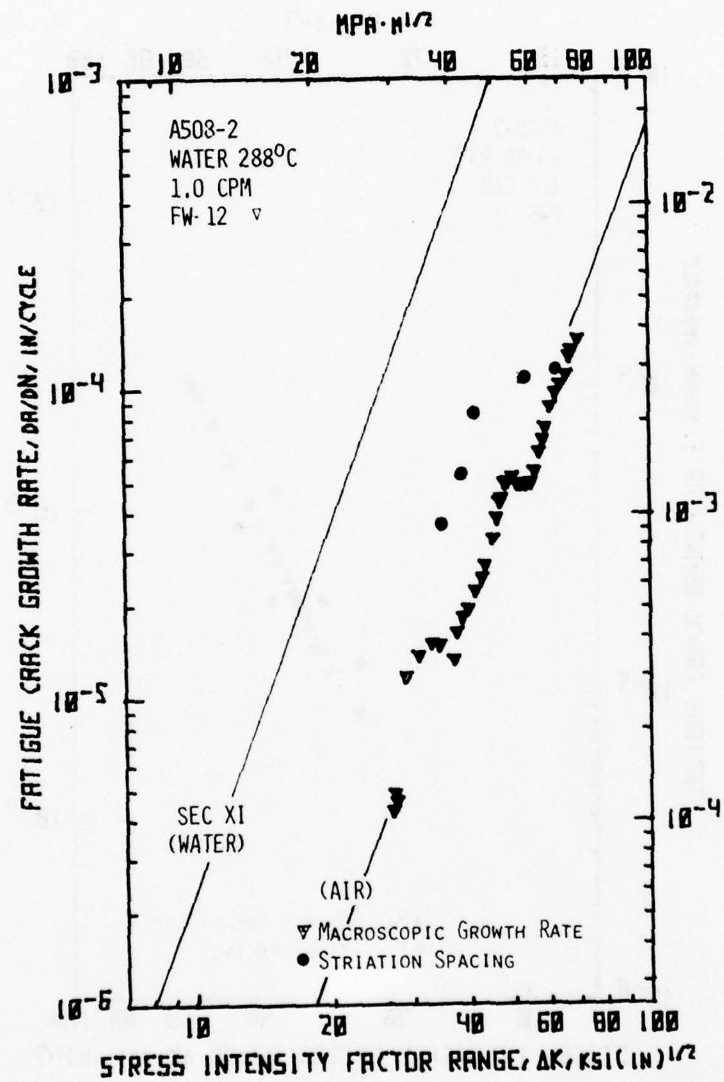

Fig. 8 — Comparison of macroscopic growth rate and striation spacing of A508 Class 2 forging steel fatigue tested at 288°C (550°F) in water

The fracture surface of a third A508 Class 2 CT specimen (Spec. FW-15 illustrated in Fig. 9), tested at 288°C (550°F) in water, was also examined with the SEM to study the fatigue striations. Especially at low values of crack length (corresponding to low ΔK values in the range of about 33 MPa√m (30 ksi √in.), the markings were both poorly resolved and discontinuous. At higher values of ΔK, the resolution was better and the discontinuity in the markings was less. The crack growth rate obtained from the striation spacings is shown in Fig. 10. For this specimen it was difficult to obtain the crack growth rate from the striation spacings. As with Specimen FW-12 (Fig. 8), the consistency between the crack growth rate computed from the SEM measurements and the corresponding macroscopic crack growth rate improved with increasing ΔK.

From the SEM study it appears that striation markings can be resolved for crack growth rates as low as 2×10^{-4} mm/cycle (8×10^{-6} in./cycle). In the event that crack growth rates as small as these cannot be easily determined macroscopically, they can be determined from the fatigue striation spacings. However, these low crack growth rate values could have significant experimental errors due to the diminished ability to resolve the corresponding striation markings. Investigations are continuing to verify this projection and to determine the lower limit of resolution by the technique.

In addition to the SEM studies described above, the method of determining the crack growth rate from beach marks, used to validate the macroscopic growth rate, was investigated with Specimens FW-15 (Fig. 9). Only one-half of the specimen is illustrated in Fig. 9 due to the sectioning required for the SEM measurements. However, the entire surface was used in the beach mark measurements. Six separate beach marks were introduced by the method previously described. The test frequency was normally 1/2 cpm, and this was increased to 15 cpm to produce the beach marks. During the test, care was taken to place the beach marks at equal intervals of crack length and to make the width of each beach mark approximately the same. The beach marks were spaced approximately 2.5 mm (100 mils) apart and their respective width was about 0.4 mm (15 mils). The total number of cycles at the beginning and at the end of each beach mark was recorded.

The beach marks on the fracture surface were obscured by an oxide film. after the test, so it was necessary to clean the surface using the same oxide removal technique

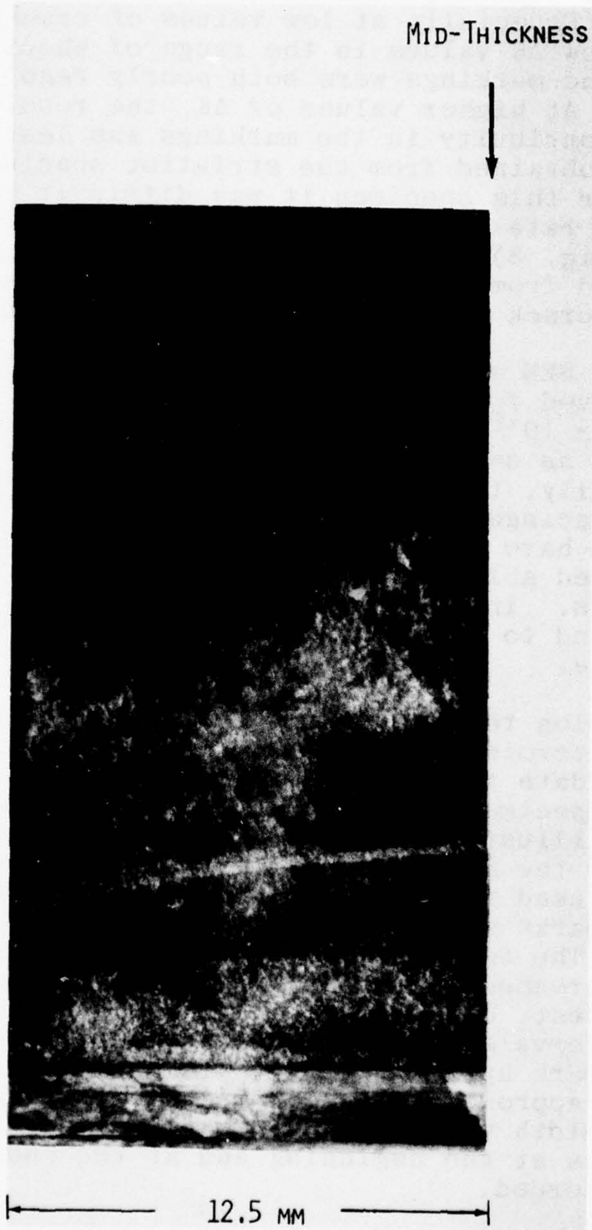

Fig. 9 — Macrograph of SEM specimen of A508 Class 2 forging steel (Specimen FW-15) tested at 288°C (550°F) in water; several beach marks can be seen

employed on the SEM specimens. It is worth nothing that the beach marks were clearly revealed after the fracture surface was lightly treated with the inhibited HCl solution to remove the oxide. Through repeated or extended solution cleaning the beach marks fade out or are completely obliterated. Figure 9 is a macrograph of the fracture surface that was later used for the SEM study. At this magnification, several beach marks are distinctly visible. The average position of each beach mark with respect to the machined notch was determined through the following method. The 25 mm (1-in.) width of the specimen along the machined notch was subdivided into eight equal parts. This subdivision is referred to as a 9-point grid (7 inner points plus the two edges). For a given beach mark, its distance to a corresponding point in the machined notch was measured with a traveling microscope. The average value of each set of nine distances for a given beach mark to the machined notch was assumed to be the crack length for the corresponding number of cycles. Thus, 12 different crack lengths (2 for each beach mark) were thus obtained. The above data are summarized in Table I. The corresponding crack growth rates were determined from the curve of crack length vs number of cycles and correlated with crack tip stress intensity factor, ΔK. The procedure used in analyzing the data was identical to that used in obtaining the crack growth rates with the compliance method. Figure 10 shows that there is very good agreement between the crack growth rate computed from the beach marks and the corresponding macroscopic crack growth rate.

SUMMARY AND CONCLUSIONS

The results of the present study can be summarized as follows:

(1) Generally, good agreement was found between crack growth rates computed from the striation spacings and the corresponding macroscopic rates; the growth rates from the striation spacings were within a factor of 2 to 3 of the macroscopic rates.

(2) The striation spacings appear to yield higher rates.

(3) From the SEM study, striation markings can be resolved for growth rates as low as 2×10^{-4} mm/cycle ($\sim 8 \times 10^{-6}$ in./cycle).

(4) For this material, the following difficulties are encountered in determining the crack growth rate by the fatigue striation method:

Table I

Summary of Data Used to Compute the Crack Growth Rate from the Beach Marks - Compact Tension Specimen Tested at 288°C (550°F) in Water

Beach Mark	Grid position along the width of the specimen; distance from machined notch in inches									Ave. Value in in.	Corresponding Number of Cycles
	1	2	3	4	5	6	7	8	9		
BM1-beginning	0.128	0.174	0.194	0.208	0.216	0.213	0.219	0.203	0.177	0.192	10,860
BM1-end	0.144	0.191	0.202	0.220	0.225	0.223	0.230	0.223	0.187	0.205	12,588
BM2-beginning	0.252	0.307	0.327	0.354	0.360	0.361	0.362	0.345	0.315	0.331	21,604
BM2-end	0.274	0.320	0.345	0.380	0.377	0.378	0.373	0.362	0.331	0.349	22,632
BM3-beginning	0.362	0.419	0.441	0.453	0.472	0.475	0.472	0.451	0.413	0.440	27,418
BM3-end	0.385	0.424	0.454	0.467	0.486	0.485	0.480	0.470	0.437	0.454	27,952
BM4-beginning	0.461	0.515	0.532	0.551	0.566	0.570	0.570	0.547	0.539	0.539	30,800
BM4-end	0.483	0.523	0.549	0.562	0.581	0.579	0.581	0.563	0.552	0.553	31,100
BM5-beginning	0.580	0.620	0.653	0.663	0.658	0.649	0.654	0.648	0.650	0.642	33,244
BM5-end	0.600	0.640	0.664	0.668	0.675	0.670	0.677	0.666	0.663	0.658	33,444
BM6-beginning	0.707	0.762	0.812	0.822	0.828	0.810	0.808	0.801	0.777	0.791	34,537
BM6-end	0.723	0.787	0.839	0.836	0.848	0.839	0.841	0.826	0.793	0.815	34,587

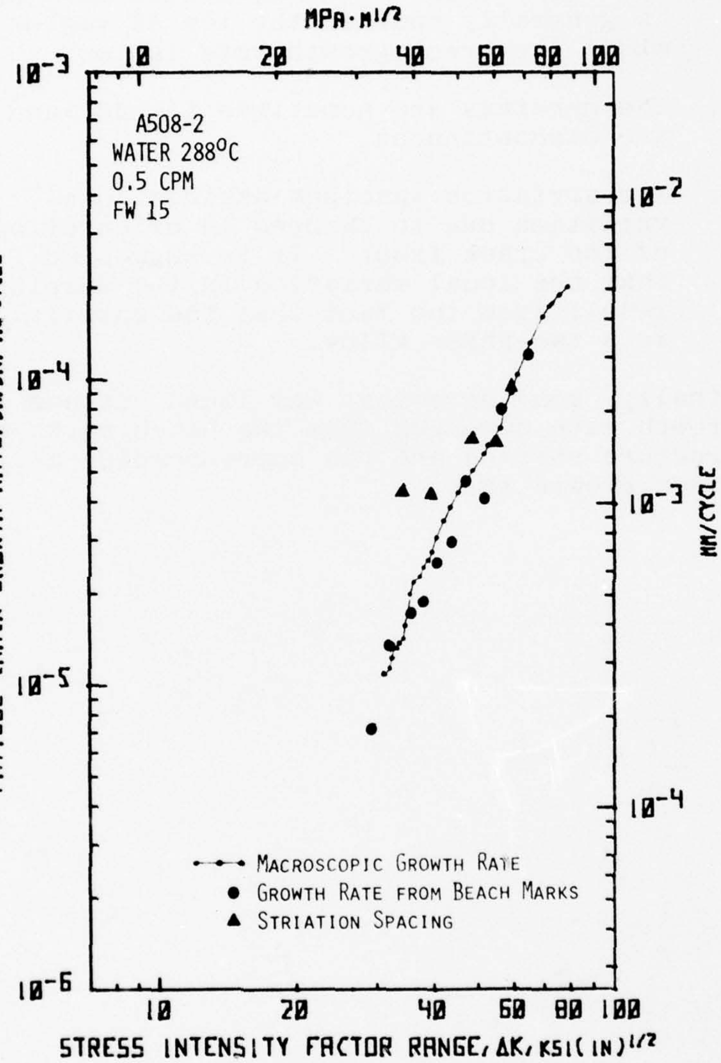

Fig. 10 — Comparison of macroscopic crack growth rate; crack growth rate obtained from the beach marks and crack growth rate from striation spacing of A508 Class 2 forging steel tested at 288°C (550°F) in water

a. Oxidation of the fracture surface can obscure or completely obliterate the striations.

b. The resolution of the striation markings is generally poor in the low ΔK region where the crack growth rate is low.

c. The markings are sometimes ill-defined and discontinuous.

d. The striation spacings exhibit local variation due to changes in orientation of the crack front. It is suggested that the local variation in the spacings result from the fact that the material is a two-phase alloy.

(5) Finally, good agreement was found between the crack growth rate computed from the beach marks on the fracture surface and the corresponding macroscopic crack growth rate.

II. RADIATION SENSITIVITY AND POSTIRRADIATION PROPERTIES RECOVERY

A. IARAR Program and Its Objective

J. R. Hawthorne, H. E. Watson and F. J. Loss

BACKGROUND

Reactor pressure vessel steels and welds currently must exhibit a minimum Charpy-V (C_v) upper shelf energy of 68J (50 ft-lb) to satisfy the ASME Code (Section III) and the Code of Federal Regulations (10CFR50). For current reactor vessel construction, there is little concern that this requirement can be met over projected reactor vessel lifetimes. However, for certain older reactor vessels which did not have the benefit of current technology, a combination of detrimental factors exist which may bar meeting this requirement after some period of extended vessel operation. These factors include the tendency of neutron radiation exposure to progressively reduce upper shelf toughness, the influence of steel impurities toward enhancing apparent radiation sensitivity and the relatively low initial (as-fabricated) upper shelf levels of the steels. Specifically, high copper impurities (>0.20%) are present to increase radiation embrittlement tendencies. At the same time, upper shelf levels are on the order of 81 to 95J (60 to 70 ft-lb) which allow only small margins for radiation shelf reductions. (In contrast, the problem has been solved in current production by specifying limits on copper, phosphorus, and sulfur impurities of 0.10, 0.012, and 0.015% (max.), respectively, and by the use of improved steelmaking and welding practices to give higher as-fabricated upper shelf levels [>122J, 90 ft-lb].)

In the event that upper shelf levels are reduced to (or below) 68J, two options exists as alternates to system shutdown. One option is the application of postirradiation heat treatment (annealing) for the relief of the radiation-induced embrittlement. Such an option was employed effectively for the Army SM-1A nuclear reactor vessel. The second option centers on an essentially volumetric examination of the beltline region of the vessel and the performance of a fracture mechanics analysis which conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of adequate margins of safety for continued operations.

In those cases where the annealing option is exercised, detailed knowledge of vessel re-irradiation behavior will be required. Moreover, it is conceivable that, for some plants, application of the annealing treatment may be necessary more than once over the system lifetime to maintain the minimum

required toughness in the vessel. Accordingly, detailed information on cyclic irradiation-annealing behavior will be required. Unfortunately, a paucity of data exists for either irradiation-annealing procedure for the steel plates and weld types of interest. The IARAR program was conceived by NRL in conjunction with NRC to gain an insight into steel performance under both operational schemes.

IARAR PROGRAM DESCRIPTION

The program initials, IARAR, stand for "Irradiate, Anneal, Reirradiate, (re)Anneal and Reirradiate," respectively. Consistent with this designation, the intent is to investigate material performance under two full annealing and reirradiation cycles. Additional features of the experimental plan are: (1) the development of material performance data at the end of each phase of the irradiation-annealing sequence; (2) an irradiation temperature of 288°C (550°F); and (3) the investigation of two postirradiation heat treatment options: 343°C (650°F) annealing and 399°C (750°F) annealing. The time period for annealing was set at 168 hours as an optimum time period based on the findings of earlier laboratory studies.

The first heat treatment option of 343°C (650°F) annealing represents the use of nuclear or pump heating to attain the requisite temperature on the vessel. The reactor coolant would be retained and, more importantly, core internals could be left in place with this option. The practical upper limit in temperature for this option is 343°C (650°F) according to a survey made of reactor vendors.

The second option of 399°C (750°F) annealing represents the use of auxiliary heaters to bring the vessel (or selected components thereof) to temperature. While this option has the capability for achieving greater embrittlement relief by virtue of a higher temperature, the removal of core internals (as well as the coolant) is necessary. Under this option, annealing at somewhat higher temperatures than 399°C appears a possibility within current reactor capabilities; however, the mechanics of a higher temperature anneal may prove too difficult in most cases.

Two A533-B submerged arc welds and one A302-B modified steel plate were selected for the investigation. All materials were produced commercially. Copper contents of the materials are 0.35, 0.35, and 0.22% Cu, respectively. Additional chemical composition information and heat treatment particulars are given in Table II. Welds V84 and V86, although similar in composition and heat treatments, were specially selected for their difference in C_v upper shelf energy levels (145 vs 98J or 107 vs 72 ft-lb). The difference stems largely

TABLE II

CHEMICAL COMPOSITIONS OF IARAR PROGRAM WELDS AND PLATE

MATERIAL	CODE		Weight (%)									
			Cu	C	Mn	Si	P	S	Ni	Mo	V	Cr
S/A Weld 1 (Linde 1092 flux)	V84	(a)	.35	.14	1.56	.14	.013	.011	.62	.53	.002	.03
S/A Weld 2 (Linde 80 flux)	V86	(a)	.35	.08	1.60	.55	.016	.013	.69	.40	.006	-
Plate 1 (A302-B Mod.)	V85	(a)	.22	.22	1.50	.27	.013	.020	.53	.46	-	-
		(b)	.24	.22	1.45	.30	.011	.014	.52	.48	-	-

<u>Heat Treatment</u>

V84 Stress relief annealed 593 to 621°C - 50 hr, furnace cooled to 316°C at 6°C/hr, air cooled.

V86 Stress relief annealed 621°C - 40 hr, furnace cooled to 316°C.

V85 843 to 899°C - 4 hr, water quenched
663°C - 4 hr, air cooled
621°C - 40 hr, furnace cooled to 316°C

(a) Vendor determination
(b) Lukens Steel Company determination

from the use of different welding fluxes (Table II). Material notch ductility and fracture toughness (K_J) will be established using standard C_v specimens and 2.5 cm (1-in.) thick compact tension (CT) specimens, respectively. The CT specimens have a crack length to width ratio (a/W) of 0.6 and are to be tested in a static mode. Primary attention will be given to upper shelf behavior under IARAR conditions; transition temperature trends will be explored as specimen numbers permit.

The radiation experiment test matrix for the program is shown in Table III. All told, fourteen experiments are involved. The matrix as presently constructed will provide full information on both A533-B welds; information for the A302-B modified plate will be developed only through Experiment No. 2. For the first phase radiation exposure, the target fluence has been set at 1 x 10^{19} n/cm^2 >1 MeV and was purposely chosen for its approximate correspondence to the knee of the radiation trend curve of transition temperature increase versus fluence at 288^{o}C (550^{o}F). For the subsequent radiation exposures, target fluence levels have not yet been selected pending the outcome of exploratory radiation and annealing tests, i.e., Experiment No. 2. Fluences for reirradiation exposures will not be as high as the initial fluence, however, since full recovery is not expected with either the 343^{o}C or 399^{o} postirradiation heat treatments.

PROGRAM STATUS

Program progress includes the completion of all irradiation, annealing and reirradiation operations for Experiments 1 and 2; postirradiation testing of both experiments has commenced. Results from these initial explorary experiments should be available in time for the next report of progress. Additional progress includes the commencement of radiation operations on Experiments 3A through 3C and Experiments 4A through 4C. Because of the time involved in irradiation and annealing operations, results in this case are not expected until 1978.

TABLE III

RADIATION EXPERIMENT MATRIX

288°C (550°F) Irradiation

Experiment Number	Specimen Types	Designation	Objective
1	C_v	IA	Explore recovery by 343 and 399°C (650 and 750°F) annealing
2 A,B,C	C_v	IAR	Explore reirradiation response of all three materials
3A through 3E	CT, C_v	I through IARAR	Determine IARAR performance of Weld 1
4A through 4E	CT, C_v	I through IARAR	Determine IARAR performance of Weld 2

B. Effect of Submerged Arc Welding on the Notch Toughness of Low Upper Shelf A302-B Plate

J. R. Hawthorne

BACKGROUND

An appreciable body of information has been developed on the Charpy-V (C_V) notch ductility of weld heat affected zones (HAZ) for reactor vessel steels having high (>122J, 90 ft-lb) initial C_V upper shelf levels. On the other hand, little data exist on HAZ properties for steels with low upper shelf levels (<88J, 65 ft-lb). A present concern is whether or not welding these steels produces a HAZ having a lower upper shelf level than the base material. Since low upper shelf plates were employed in some early reactor vessel construction, resolution of this uncertainty has practical importance, especially in view of current code requirements on minimum C_V notch ductility.

PROGRESS

The effect of high heat input, submerged arc welding on the C_V notch ductility of an A302-B plate having a low upper shelf level has been explored. The study employed simulated weld HAZ specimens prepared* in a Gleeble apparatus. The specimen blanks were taken from the parent plate at its one quarter thickness (1/4T) location and in the transverse (TL) orientation. The simulated weld thermal cycle depicted the welding of a 10.2 cm (4-in.) thick carbon steel plate using a heat input of 39.4 kilojoules/cm (100 kilojoules/in.) and a preheat of 177°C (350°F). The peak temperature of the thermal cycle was 1316°C (2400°F). A 621°C (1150°F)-32 hour stress relief anneal was applied following the weld simulation.

The C_V data developed for the HAZ simulation (open triangles) are compared to the performance of the unwelded plate (see 1/4T location, TL orientation) in Fig. 11. Significantly, the HAZ thermal cycle produced a 25 percent lower C_V upper shelf (49J, 36 ft-lb) than that measured for the parent plate (65J, 48 ft-lb). The concern over HAZ upper shelf properties for the case of low shelf plates thus appears justified. In Fig. 11, it is noted that the thermal cycle also produced a lowering of the C_V 41J (30 ft-lb) transition temperature. This beneficial effect, however, is masked by the degradation of upper shelf notch ductility.

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*Courtesy U. S. Steel Applied Research Laboratory, (Dr. W. D. Forgeng)

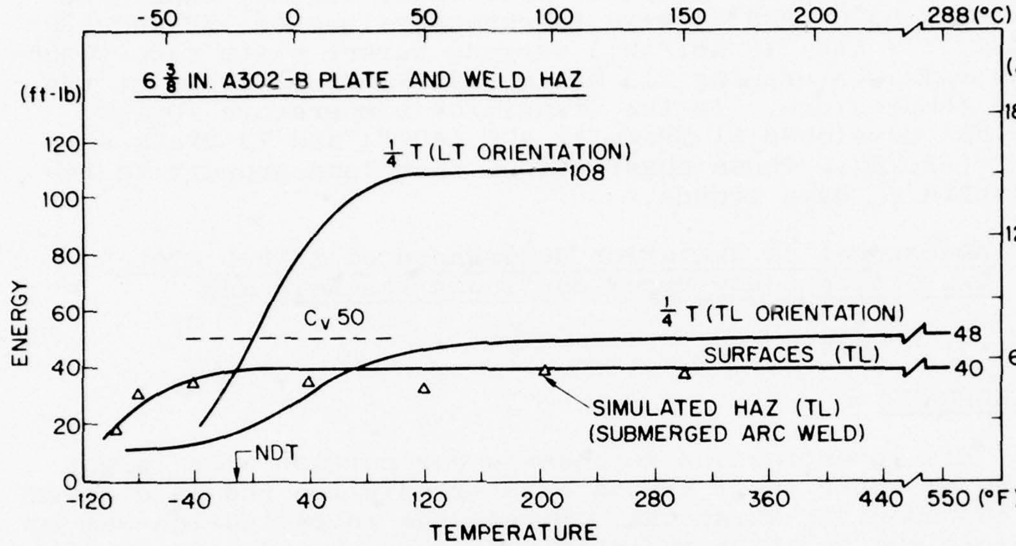

Fig. 11 — Charpy-V notch ductility of the simulated weld HAZ (open triangles) compared to the A302-B parent plate (curves). HAZ specimen blanks were removed from the plate quarter thickness location in the transverse (TL) test orientation. Trend curves for the plate surface location (TL) and for the quarter thickness location (LT orientation) are also shown for reference.

In further test of the HAZ, selected specimens were fatigue precracked and tested in accordance with Electric Power Research Institute (EPRI) procedures for measuring fracture toughness. Dynamic fracture toughness (K_J) determinations were by the J-integral method and were based on energy absorbed to maximum load* corrected for specimen and machine compliance. In the upper shelf regime, duplicate tests at 93°C (200°F) gave toughness values of 129 and 130 MPa√m (117 and 118 ksi√in.) whereas parent plate gave toughness values averaging 213 MPa√m (194 ksi√in.) at about the same temperature. In the transition temperature regime, the HAZ developed 91 MPa√m at 4°C (40°F) and 73 MPa√m at -34°C (-30°F). These observations thus lend support to the standard C_v data trends.

C. Assessment of Radiation Resistance of A302-B Plate Exhibiting a Low Upper Shelf and Its Weld HAZ

J. R. Hawthorne

BACKGROUND

This investigation represents one portion of a larger study of upper shelf trends with irradiation and is directed to two specific questions. First, the investigation was to explore the relative radiation effect on C_v properties of strong (LT, longitudinal) versus weak (TL, transverse) test orientations for the case where preirradiation upper shelf levels differ significantly and where the upper shelf level of the TL orientation is low. Secondly, the investigation was to compare the relative radiation effect on C_v properties of the TL orientation versus the weld heat affected zone (same test orientation) produced by submerged arc welding. Relative to the first question, experimental data trends (7) suggest that the radiation degradation of upper shelf level is greater for that orientation having the greater initial upper shelf, i.e., LT orientation. Data applicable to the second question are not known.

PROGRESS

Standard and fatigue precracked C_v specimens representing the LT and TL orientation of an A302-B steel plate and a simulated weld heat affected zone (HAZ) have been irradiated at 288°C (550°F) to 2.1 x 10^{19} n/cm^2 >1 MeV. Specifications

*J-integral determinations on the basis of maximum load imply the absence of stable (rising load) crack extension. This is normally not the case for the type of specimen employed. Thus, the K_J values determined would tend to overestimate the K_J at crack initiation.

for the plate, purchased earlier (8), were intended to maximize the difference indirectional properties; that is, the specifications called for the use of a minimum of cross rolling in the processing of the original ingot and slab to plate. The weld HAZ specimens were produced to specifications and procedures described in the preceding section. To conserve material stock, the weld HAZ specimens (blanks) were taken from the 1/4T location while the plate specimens were taken from the adjoining 3/4T location. As will be noted below, the differences in C_V properties between the two plate locations were negligible with regard to the upper shelf and was small with regard to transition behavior. The chemical composition of the plate* (and HAZ) is as follows:

C	Mn	P	S	Si	Ni	Cr	Mo	Cu
.21	1.46	.010	.021	.24	.23	.06	.54	.059

Because the copper and phosphorus contents of the plate are relatively low, large radiation-induced changes in notch ductility were not expected. By the same token, the plate provided the opportunity to demonstrate for A302-B, improved radiation resistance through the restriction of these two elements. A commercial scale demonstration test for A533-B steel is already available (9).

The results from standard C_V tests of the plate and HAZ are summarized in Figs. 12 and 13, respectively. Several observations are permitted as follows:

(1) High radiation resistance in terms of notch ductility is demonstrated by each of the materials. In the case of the plate, the results serve to validate for A302-B the NRL-developed guidelines for commercial production of improved (radiation-resistant) steel.

(2) The effect of radiation on upper shelf level appears to be the same for both plate LT and TL orientations; similarly, the effect of radiation on transition temperature was comparable for both orientations.

(3) The results demonstrate that radiation can produce a significant transition temperature elevation without reducing upper shelf energy level in the materials.

(4) The HAZ appears much more sensitive to radiation than the plate in terms of transition temperature elevation; however, it is noted further that the HAZ transition temperature was not elevated above that of the parent plate by the radiation treatment.

*Courtesy Babcock and Wilcox Company

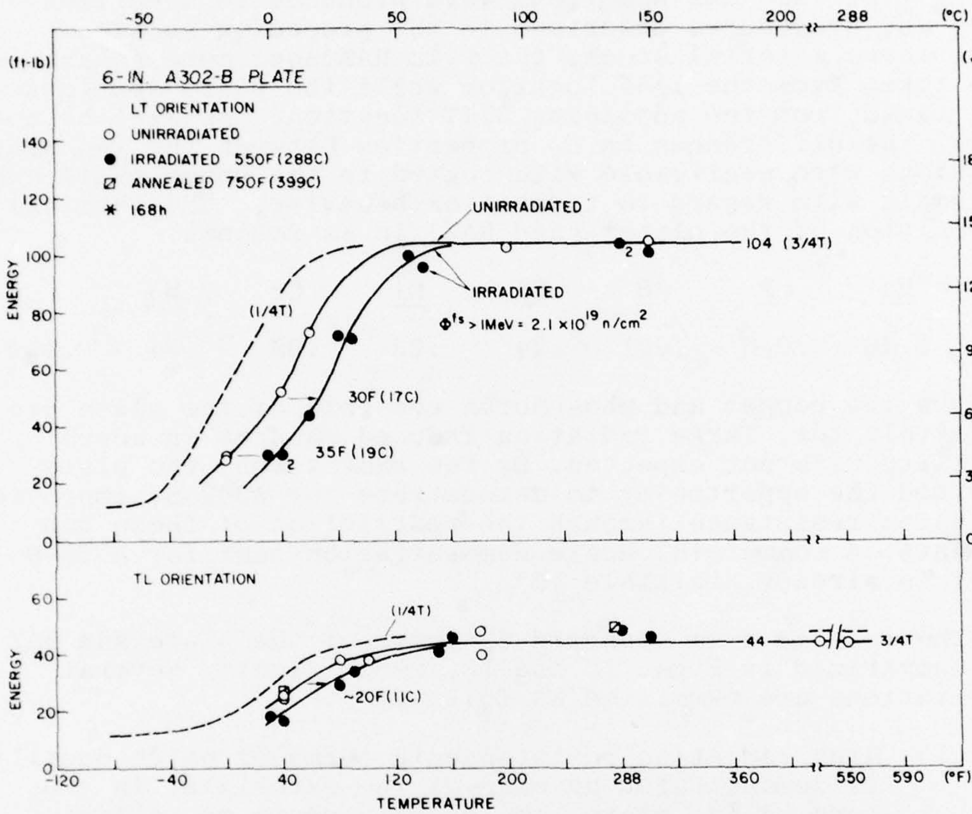

Fig. 12 — Charpy-V notch ductility of the A302-B plate in longitudinal (upper graph) and transverse (lower graph) test orientations before and after 288°C irradiation. A datum for the 399°C (750°F) postirradiation heat treated condition is also shown. The dashed curves describe the performance of the plate 1/4T location from which the HAZ specimens (see Fig. 13) were taken.

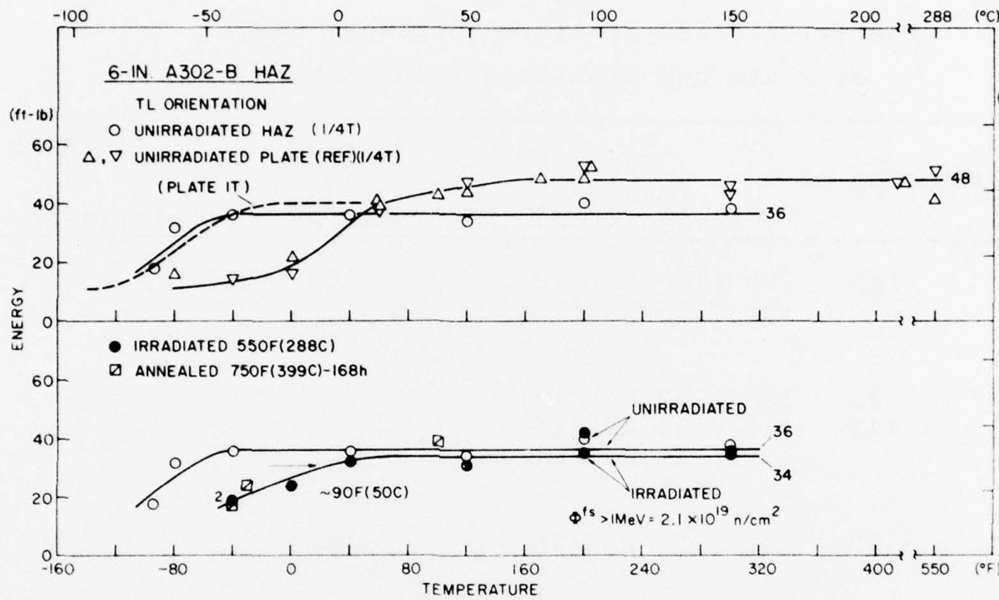

Fig. 13 — Charpy-V notch ductility of the simulated weld HAZ in the transverse test orientation. The upper graph shows data for the HAZ relative to that of the parent plate at the same (1/4T) test location. The lower graph compares the properties of the HAZ before and after the irradiation.

TABLE IV

Postirradiation Fracture Toughness (K_J) of Plate and Simulated Weld HAZ

Material	Deg C	K_J (MPa$\sqrt{m}$) (Irradiated)	Average K_J (Irradiated)	Average K_J (Unirradiated)
Plate (LT)	143	185 (168 ksi$\sqrt{in.}$) 192 (175 ksi$\sqrt{in.}$)	189	-a-
Plate (TL)	93 143	156 (142 ksi$\sqrt{in.}$) 141 (128 ksi$\sqrt{in.}$) 158 (144 ksi$\sqrt{in.}$)	152	213
HAZ	93	143 (130 ksi$\sqrt{in.}$)	(143)	130

[a]Not available

Overall, the findings indicate the need for more detailed investigations of plate versus weld HAZ postirradiation properties and property changes for more radiation-sensitive plate versions.

Preliminary fracture toughness (K_J) determinations for the materials, corresponding to the postirradiation C_V upper shelf, are given in Table IV. As expected from the C_V data, the HAZ did not exhibit a reduction in K_J level with irradiation. On the other hand, a large reduction in average K_J is noted for the TL orientation of the plate. This apparent reduction may be associated with the use of the maximum load criteria for the determination of K_J. Nonetheless, this anomaly bears clarification. Additional fracture toughness tests currently are in progress.

III. SURVEY OF FOREIGN RESEARCH RELATED TO THE SAFETY OF LIGHT WATER REACTORS

A. Visits to Japanese Research Organizations(15-29 Apr 1977)

L. E. Steele

INTRODUCTION

The main purpose of the visit was to attend the Third International Conference on Pressure Vessel Technology in Tokyo and to participate in related discussions with various industrial concerns and research organizations in Japan. In addition, two invited lectures were given, one at the Japan Atomic Energy Research Institute and the other at the University of Tokyo at a post-conference session. Visits to the several industrial or research concerns produced discussions on subjects of direct interest to our task for NRC, the structural integrity of water reactor pressure boundary components. The highlights of these visits are summarized as they pertain to tasks underway for NRC at NRL.

VISIT TO JAPAN ATOMIC ENERGY RESEARCH INSTITUTE (JAERI)

At this site an invited lecture was presented entitled, "Assuring Reactor Pressure Vessel Reliability Considering the Neutron Environment." Discussions included a review of work being carried out there on fatigue, on structural mechanics and on radiation damage. The emphasis at JAERI is primarily on the gas-cooled system and the breeder so that much of the work is not of direct interest to the NRC-NRL program. Nevertheless, there was a meeting with individuals, including Dr. Kondo, Dr. Miyazona, and Dr. Oku, who have direct interest in the behavior of materials for reactor pressure components. One of the most useful discussions was that of a review with collaborators of Dr. Kondo from Tokyo University who reviewed some work on fatigue crack propagation suggesting more rapid crack growth in the heat affected zone than in the base forging or plate. (A detailed list of reports obtained at JAERI is included as Appendix A . If additional information is needed relative to this visit, it may be obtained by personal request to the author).

Third International Conference on Pressure Vessel Technology

No detailed report is provided of this conference as NRC has available volumes containing the papers presented at the conference and an NRC representative was present at the meeting. However, the author has prepared a brief analysis of the conference for publication in the Scientific Bulletin of the Office of Naval Research, Tokyo, which may be of interest to the reader (10).

Nippon Steel Research Laboratory, Kanagawa-Ken, Japan

At this site, interest was focused on the large testing which clearly took the form of evaluations of Nippon product line items to simulate geometry and loading in service. These included ship hull materials, bridge structural steels, and other large structures including chemical pressure vessels and components for nuclear power plants. The most interesting outcome of this meeting followed an informal lecture presented by the author entitled, "A Review of Some Critical Factors in Steelmaking to Support Nuclear Power." It was pointed out by Nippon Steel representatives that control of composition was a normal practice in their laboratory and had been for many years. In addition, they wished to discuss the techniques of welding which are being developed in their laboratory for customers. An item of particular interest was a technique for narrow gap welding which seemed to provide a very fine grained structure using an automated continuous weld process. However, I pointed out that as they were using copper coated electrode material, it would be important to know that they did not have segregation of the copper. They indicated that the very low amount of copper on the electrode would produce .06 copper in the weld, and I indicated that this is not a problem in the irradiation field, but if it were to segregate along one of the weld boundaries or in the heat affected zone or on the fusion line it could be a problem. They agreed to make a careful analysis of the distribution of copper in the new weld type.

Ad Hoc Meeting of the Working Group on Reliability of Reactor Pressure Components

This meeting was called at the request of the Japanese representative to the IAEA Working Group for the purpose of finalizing some plans for materials to be included in an IAEA coordinated program entitled, "Analysis of the Behavior of Advanced Reactor Pressure Vessel Steels Under Neutron Irradiation." Planning was for the benefit of the Japanese participants in particular but reached some important conclusions which may bear on irradiation damage studies of the United States. Major Japanese suppliers and research

organizations contributed. These include: Japan Steel (heavy forgings), Kobe Steel (weld material), Nippon Steel (heavy plate), Mitsubishi (welding), and ISES (coordination). It was agreed that Japan would furnish plate and forging enough for six major investigators and welds for at least half as many. In addition, it is expected that an European equivalent will be provided and that irradiations would be undertaken in several countries. Four have already agreed; these include: Czechoslovakia, Federal Republic of Germany, Japan, and the USA. Additional information will be provided after a meeting of the coordinated program participants in October in Vienna.

Japan Steel Works (JSW)

During this visit the author presented a paper entitled, "Review of Radiation Damage of Steel," at the request of host, Deputy Director Dr. Onodera. This formed the basis for discussion of radiation damage and steel quality. It was pointed out by the Japanese that they had provided steels of certain quality (composition) levels as indicated in Table V by the time period. Since 1971, control of copper, phosphorus, vanadium, and sulfur contents has been standard practice at JSW. A tour of the facilities covering melting, forging, heat treatment, and machining was very enlightening in that it was possible to see the integral nozzle type shell courses for vessels being prepared for PWR contractors in Germany (KWU) and Belgium (Cockerill). This was a fascinating demonstration of large forging capacity. Japan Steel Works apparently has the largest capacity in the free world for melting and handling steel ingots and this makes it normal for them to undertake the new designs since the advantages of the integral nozzle can only be obtained by very massive forgings and careful machining. These were shown along with the merits in terms of stresses, capacity for nondestructive examination of welds as well as the possibility for avoiding through-wall weld penetrations in the primary pressure vessel by having integral nozzle offsets for easy welding of piping. Another observation at Japan Steel Works was the use of a miniature specimen which was called a J-Integral specimen. I attempted to obtain information on this but have only a promise that they will provide a paper on the specimen and the technique for measuring once it is released for publication which should be during the period of Summer 1977.

CONCLUSIONS

In each of the sites visited in Japan there was a genuine interest in the work of the Laboratory and particularly in regard to safety and reliability research for the light

TABLE V

LIMITS ON RESIDUAL ELEMENTS FOR PRESSURE VESSEL STEELS AND WELDS

Element	USA Standards Maximum Permitted (Non-Nuclear) (A533-B) %	USA Standards Maximum Permitted (Nuclear)[a] %	Japan Steel Works 1966-1970	Japan Steel Works 1971 & Later
Copper	Not specified	0.10	0.12	<0.06-0.08
Phosphorus	0.035	0.012	0.012	<0.008
Vanadium	Not specified	0.05	0.01	<0.01
Sulfur	0.040	0.015	0.012	<0.008

[a]Plate (check analysis), forging (check analysis), weld (as deposited).

water reactor. Related to this was an interest in obtaining information from us which was matched by a willingness on their part to provide reports and information as evidenced by the long list of reports provided at each of the sites, particularly JAERI, where thirteen reports were obtained. These included some from cooperative studies from Tohoku University. For easy reference a list of those reports is provided in Appendix A.

B. European Research on Fatigue Crack Propagation in Water

H. E. Watson

BACKGROUND

On 18-19 May 1977, a conference entitled, "The Influence of Environment on Fatigue," was held at the Institute of Mechanical Engineers, London, England. While in Europe to attend this conference, visits were made to several laboratories in the United Kingdom, France, and Germany to discuss the status of fatigue crack propagation (FCP) research relating to a reactor water environment and to compare results with those obtained in the USA.

OBSERVATIONS

Environmental FCP research is of primary interest in all three countries. The lack of equipment required to conduct tests has presented problems in the past; however, several laboratories have just activated new test systems or will be doing so within a few months. Autoclave FCP tests using low alloy carbon steels are underway at Harwell and the Central Electricity Research Laboratories in the United Kingdom, at Creusot-Loire in France, and at Kraftwerk Union in West Germany. Environmental FCP tests at room temperature are underway at the United Kingdom Atomic Energy Authority in Risley.

The objectives of the European FCP research programs related to water reactors closely parallel those of the USA. Specifically, the performance of A508 and A533 plate (or similar materials) as well as welds and heat affected zones in both PWR and BWR systems is being analyzed from a FCP standpoint. Test variables include: cyclic frequency, R-ratio, coolant chemistry, temperature, and stress wave shape.

Some preliminary conclusions have been reached as a result of the European research effort in FCP in low alloy carbon steels. Those discussed during this trip are as follows:

(1) At low ΔK, effect of frequency is small.

(2) The effect of reducing cyclic frequency is most prominent at ΔK = 20 MPa$\sqrt{m}$.

(3) Generally, reducing cyclic frequency from 1.0 hz to 0.1 hz produces a significant increase in FCP.

(4) High R-ratio tests produce data very close to the ASME Section XI water line.

(5) An effect of irradiation on FCP was observed in A508 and A533-B welds. No effect has been observed in plate material.

(6) Crack growth rates in the HAZ are higher than in reference tests on plate.

(7) Water chemistry is believed to play a significant role in FCP tests.

(8) Hydrogen has a significant effect on electrical potential at 288^{o}C (550^{o}F), thereby affecting the corrosion rate.

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9. J. R. Hawthorne, "Demonstration of Improved Radiation Embrittlement Resistance of A533-B Steel Through Control of Selected Residual Elements," NRL Report 7121, Naval Research Laboratory, Washington, D. C.,May 21, 1970.

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APPENDIX A

LIST OF REPORTS PROVIDED BY JAERI REPRESENTATIVES

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2. K. Shiraishi, A. Hishinuma, and Y. Katano, "Radiation-Generated Prismatic Loops Around Gas Bubbles in Aluminum-Lithium Alloy," Radiation Effects 21, 1974, pp. 161-164.

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